NON-PUBLIC?: N
ACCESSION #: 8712220077

LICENSEE EVENT REPORT (LER)

FACILITY NAME: North Anna Power Station, Unit 1 PAGE: 1 of 4

DOCKET NUMBER: 05000338

TITLE: Reactor Trip Generated From 5A Feedwater Heater Hi-Hi Level Signal
EVENT DATE: 11/23/87 LER #: 87-020-00 REPORT DATE: 12/15/87

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: E. Wayne Harrell, Station Manager TELEPHONE #: 703-894-5151

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SM COMPONENT: LS MANUFACTURER: M235
REPORTABLE TO NPRDS: Y
CAUSE: X SYSTEM: SB COMPONENT: TCV MANUFACTURER: C635
REPORTABLE TO NPRDS: Y
CAUSE: X SYSTEM: AA COMPONENT: ZI MANUFACTURER: I130
REPORTABLE TO NPRDS: Y
CAUSE: X SYSTEM: SJ COMPONENT: MOV MANUFACTURER: C684
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: At 0009 hours on November 23, 1987, Unit 1 tripped from 100 percent power (Mode 1). The initiating signal for this Reactor Trip was a turbine solenoid trip which resulted from a 5A feedwater heater Hi-Hi level signal. The 5A FW heater Hi-Hi level signal was generated when a level switch failed. This event is reportable pursuant to 10CFR50.73(a)(2)(iv). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

The cause of the level switch failure was fatigue failure of a spring inside the microswitch. When the spring failed the switch closed and a turbine solenoid trip signal was generated. As corrective actions, the level switch was replaced and all other feedwater heater level switches which initiate reactor trips were inspected with satisfactory results.

This event posed no significant safety implications because all safety related equipment functioned as designed and key reactor parameters stabilized following the reactor trip. Also, the reactor trip signal was generated from a secondary plant protection signal for which an actual condition did not exist. The health and safety of the general public were not affected.

(End of Abstract)

1.0 Description of Event

At 0009 hours on November 23, 1987, Unit 1 tripped from 100 percent power (Mode 1). The initiating signal for this Reactor Trip was a turbine solenoid trip above P-7 (2 of 4 power range nuclear instruments greater than 10 percent power or 1 of 2 turbine impulse chamber pressure greater than 10 percent rated turbine power). The turbine solenoid trip resulted from a 5A Feedwater (FW) heater (EIIS System Identifier SM, Component Identifier HX) Hi-Hi level signal although no actual Hi-Hi level condition existed. This event is reportable pursuant to 10CFR50.73(a)(2)(iv). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

A Hi-Hi level signal in the 5A FW heater results in closing of two level switches, LS-SD-129A or LS-SD-129C (EIIS System Identifier SM, Component Identifier LS, Vendor Identifier M235, Model No. 201EV-7806). Closure of either level switch generates a signal to trip the turbine, lock out the condensate pumps, and open the condenser vacuum breaker MOV. The Hi-Hi level signal which resulted in this reactor trip was due to the failure of LS-SD-129C. When LS-SD-129C failed, it completed the circuit for the automatic signals described above. As a result of all three condensate pumps (EIIS System Identifier SD, Component Identifier P) being tripped and locked out, the two running main FW pumps (EIIS System Identifier SJ, Component Identifier P) tripped on low suction pressure as designed. The standby main FW pump automatically started but tripped within 10 seconds due to low suction pressure.

Following the reactor trip, primary system pressure and temperature decreased to approximately 1860 psig and 520 degrees F, respectively, then quickly recovered to the normal no load values of 2235 psig and 547 degrees F. The Reactor Coolant System (RCS) temperature decrease to 520 degrees F was primarily due to the loss of the condensate and main FW pumps which required the auxiliary FW system (which started on Lo-Lo Steam Generator (S/G) level) to feed relatively cold (ambient) water from the Emergency Condensate Storage Tank into all three S/G's for an extended period of time. Running of the steam driven auxiliary FW pump (EIIS System Identifier, Component Identifier) also contributed to the cool down because the steam used to drive the pump was being extracted from the S/G's. The decrease in RCS temperature resulted in letdown isolation due to pressurizer level shrinking below 15 percent.

Plant equipment responded as expected with the following exceptions:

o Two Individual Rod Position Indicators (IRPIs) (EIIS System Identifier AA, Component Identifier ZI), G-13 on Shutdown Bank A and P-6 on Control Bank A, did not correctly indicate rod position. G-13 indicated approximately 36 steps and its rod bottom light did not illuminate. P-6 indicated approximately 20 steps but its rod bottom light did illuminate. (The problems were determined to be IRPI related and were corrected.)

o One steam dump valve (TCV-MS-1408C) (EIIS System Identifier SB, Component Identifier TCV) failed to close until the P-12 interlock (2 of 3 RCS loop temperatures less than 543 degrees F) was met. Air to TCV-MS-1408C was isolated in order of isolate the valve. (A solenoid on TCV-MS-1408C was replaced and the valve was returned to service.)

o The main FW isolation MOVs (MOV-FW-154B and C) (EIIS System identifier SJ, Component Identifier ISV) did not fully close. This condition was identified after restoring condensate feed to the S/G's and securing auxiliary FW. The S/G levels continued to increase even though the main and bypass FW regulation valves were closed. In an unsuccessful attempt to correct this condition, the main FW isolation MOVs were also closed. (Both MOVs were MOVATS tested, and returned to service following adjustment of the limit switches.)

2.0 Safety Consequences and Implications

This event posed no significant safety implications because all safety related equipment functioned as designed and key reactor parameters stabilized following the reactor trip. Also, the reactor trip signal was generated from a secondary plant protection signal for which an actual condition did not exist. The health and safety of the general public were not affected.

3.0 Cause of the Event

The cause of this event was failure of LS-SD-129C. LS-SD-129C is a normally open contact which closes when a Hi-Hi level in the 5A FW heater is present. When a spring in the microswitch experienced fatigue failure LS-SD-129C closed.

4.0 Immediate Corrective Action

As an immediate corrective action, water induction circuits were de-energized in order to start the condensate pumps and begin secondary system recovery actions.

5.0 Additional Corrective Actions

As an additional corrective action, LS-SD-129C was replaced. (The corrective actions for those components which did not function properly are mentioned in the description). Also, all feedwater heater level switches which generate reactor trips were inspected with satisfactory results.

6.0 Actions Taken to Prevent Recurrence

To prevent recurrence of this type event, an evaluation to install additional level switches will be performed. The addition of level switches would allow changing the logic requirements so that a single failure would not result in a turbine trip yet preserve the necessary secondary plant protection.

7.0 Similar Events

No previous similar events have occurred involving failed level switches causing a reactor trip signal to be generated.

8.0 Additional Information

Unit 2 was stable in Mode 1 throughout this event and was not affected.

The condenser vacuum breaker did not open upon closure of LS-SD-129C because the condenser vacuum breaker MOV was deenergized closed.

ATTACHMENT # 1 TO ANO # 8712220077 PAGE: 1 of 1

10 CFR 50.73

VEPCO VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117

December 15, 1987

U. S. Nuclear Regulatory Commission Serial No. N-87-042
Attention: Document Control Desk NO/DEQ: nih
Washington, D.C. 20555 Docket No. 50-338

License No. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. LER 87-020-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very Truly Yours,

E. Wayne Harrell
Station Manager

Enclosure

cc: U. S. Nuclear Regulatory Commission
101 Marietta Street,
Suite 2900
Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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